Controlling a DEMO reactor with a sparse set of diagnostics

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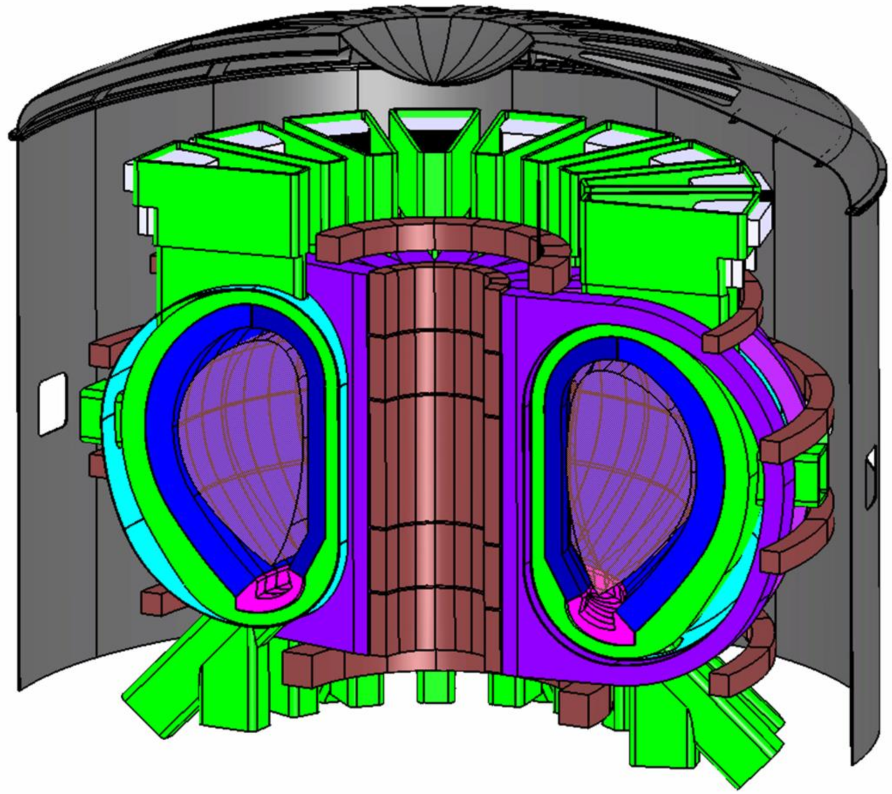
The European approach towards a DEMO fusion reactor

- DEMO as the single step between ITER and a commercial power plant
- The tokamak as baseline, stellarator as alternative option
- Main targets for DEMO:
  - significant net electricity in ~ 2050
  - Tritium self-sufficiency (TBR > 1)
  - demonstrate all relevant technologies, adequate availability (30 – 50%)
  - perspective for economic attractiveness of fusion
- Two variants under consideration:
  - pulsed conservative design as baseline → DEMO 1
  - more advanced steady state design as option → DEMO 2
  - additional variants to be considered in 2017, e.g. DN divertor



Main DEMO parameters:

- R, a ~ 9.1 m, 2.9 m
- P_{th} ~ 2 GW
- $P_{el,net}$ ~ 300…500 MW
- t_{pulse} > 2 h

Role of plasma diagnostics on DEMO

- Plasma diagnostics on DEMO will mainly/only be installed for plasma control
  - quite different from any existing fusion device and from ITER
- Main requirements for the DEMO diagnostic & control system:

1. Provide stable machine operation in compliance with safety requirements
   - Any control failure should not generate significant safety issues
   - If any active control needed for safety → more demanding requirements on control
2. Avoid machine damage, keep safe distance from all operational limits
   - no unmitigated full energy disruptions (< 1 / fpy)
   - avoid or mitigate any other strong transients or off-normal events (avoid melting of the wall)
3. Optimise reactor performance for one plasma scenario
   - minimise cost of electricity
   - maximise net electrical output power
   - maximise reactor availability
   - maximise component lifetime (minimise erosion, cyclic loads, neutron embrittlement)

DEMO control quantities and operational limits

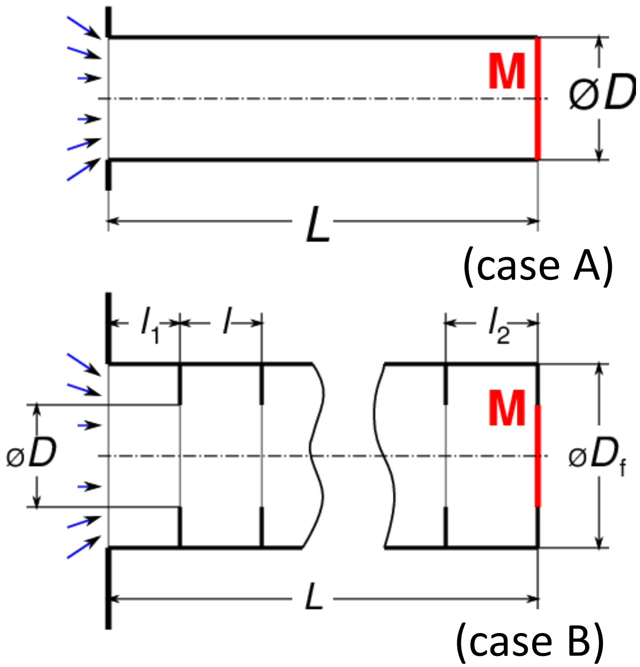
Control quantity	Operational limits
Plasma current	q_{95} limit (safety factor)
Plasma density (edge rather than core)	density limit
Plasma radiation, impurity mixture, Z_{eff}	radiation limit LH threshold
Fusion power	local wall loads (FW and div.) LH threshold
Divertor detachment and heat flux control	LH threshold local wall loads in divertor
Plasma position and shape, incl. vertical and horizontal stability	local wall loads (FW and div.) max. Δz / VDE disruption
(MHD) plasma instabilities	various (→ disruptions)
Plasma pressure	beta limit

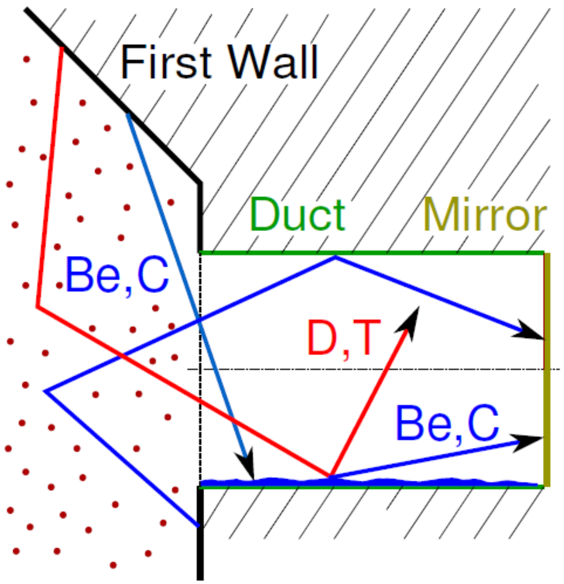
Technical requirements for the DEMO diagnostics and control system

1. Full coverage of all control needs under all foreseeable plasma conditions
   - reliable and accurate signals to be provided to the control system under all normal and off-normal conditions
   - may need redundance in terms of methods
2. High availability and reliability over long time
   - provide reliable plasma operation
   - Reliable operation of the D&C system over several full-power years (equal to blanket or divertor lifetime) without the need for longer maintenance periods
   - may need redundance in terms of number of channels
3. Minimum impact on the tritium breeding rate (TBR) and neutron shielding
   - available area and volume for in-vessel diagnostic implementation on DEMO is only a few percent of the blanket area and volume (ΔTBR < 0.04)
   - preferably use diagnostic systems with low space consumption of front-end components
4. Cost minimisation, standardisation
   - Standardisation of blanket boxes, and divertor modules with proper integration of diagnostics and actuators components (diagnostic modules)
   - Standardisation of diagnostics and actuator components
   - Preferably use proven methods and technologies (reduce risks and development effort)

Diagnostic lifetime issues: First mirror passive protection by high L/D ratio + baffles

- First mirrors are subject to erosion and deposition fluxes from the plasma side
- Simulation of flux attenuation by duct geometry via B2-Eirene modelling (V. Kotov et al., Phys. Scr. 2011)
- 2 cases with and without baffles considered
- Length over diameter ratio (L/D) as the main parameter for variation





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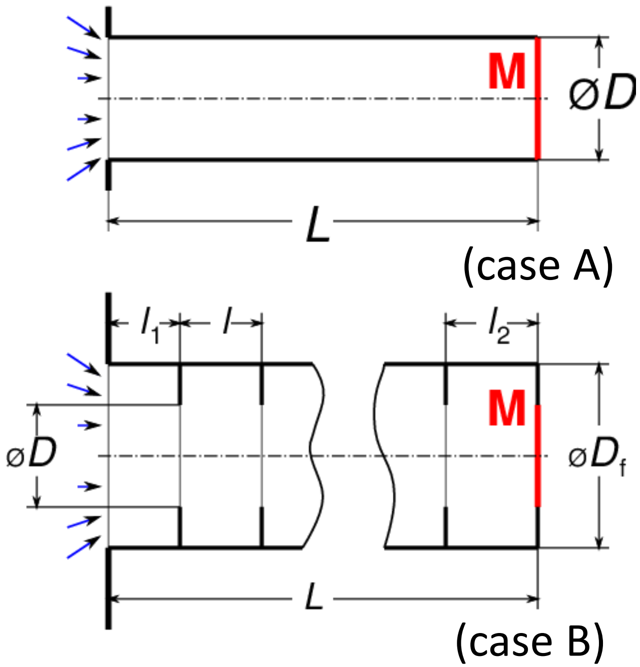


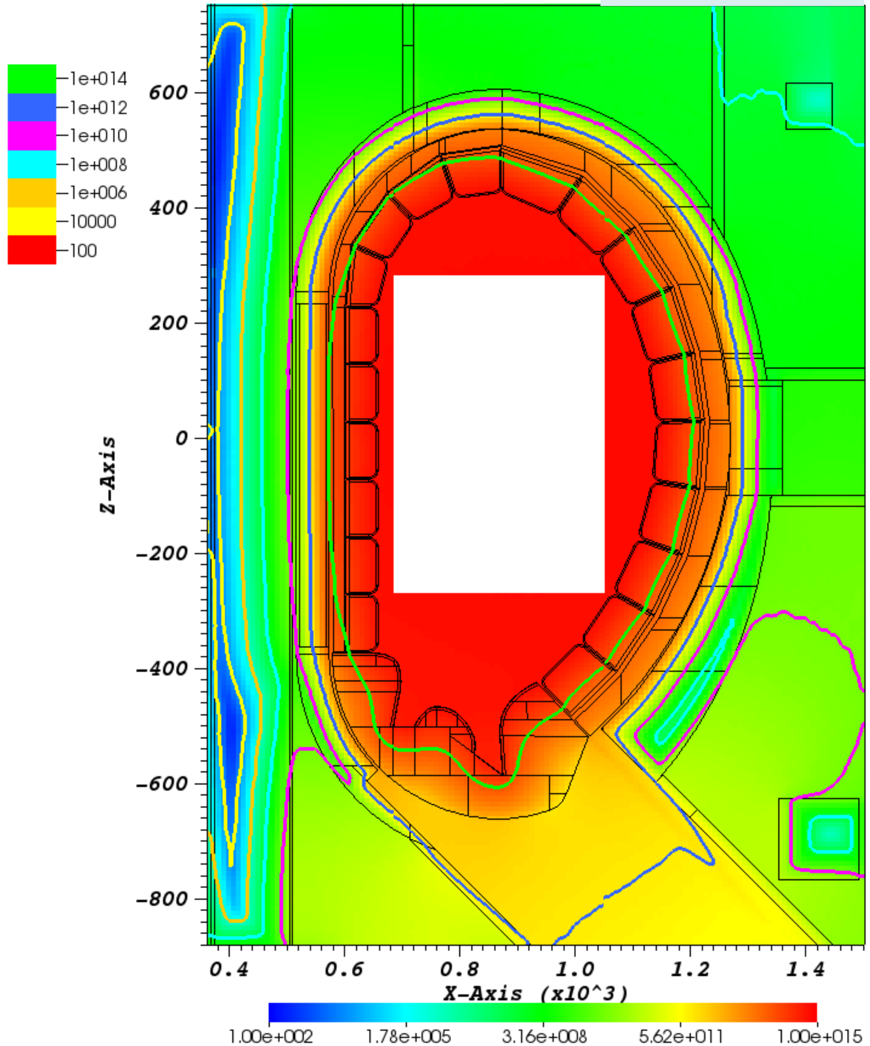
Table: Expected attenuation factors for deposition fluxes

L/D		2	5	10	20	30
R_N=0.9	A	4.0..5.0	7.9..9.6	20..23	150..190	550..720
f_{enh}=10	B	8.0..8.9	60..72	740..970	4000..5800	15000...24000
R_N=0.5	A	5.6..6.3	32..37	170..200	910..1200	3000..4300
f_{enh}=5	B	10..11	470..740	3000..6400	13000..32000	30000..95000
f_{enh}=1	B	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

First mirror lifetime studies being performed for DEMO with W first wall

Radiation issues for plasma diagnostic front end components on DEMO

DEMO neutron flux map (T. Eade, CCFE)



Typical fluences for magnetic sensors (behind the blanket manifold):

ITER:	1	GGy
	10^{20}	n/cm^2
DEMO:	1.2	GGy (gamma + n, OB)
	12	GGy (gamma + n, IB)
	10^{21}	n/cm^2 (OB)
	10^{22}	n/cm^2 (IB)

(G. Vayakis, ITER_D_26ZGC3)

(T. Eade, CCFE)

Radiation effects on diagnostic components:

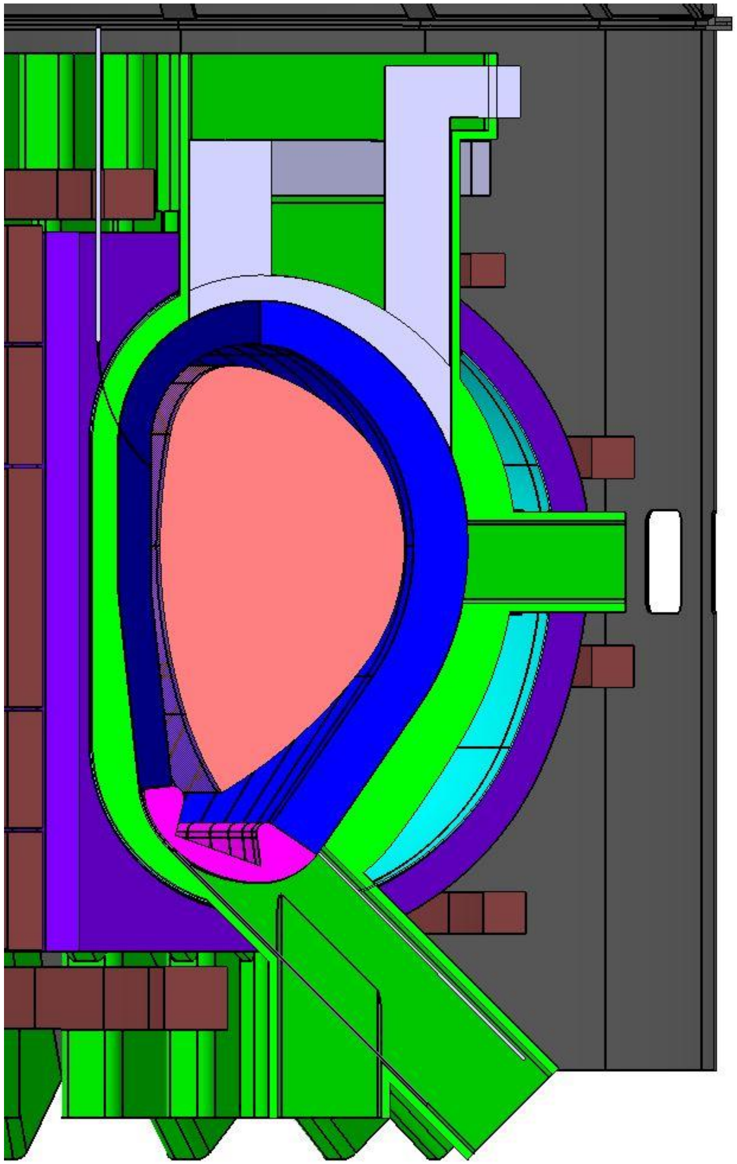
- radiation-induced conductivity
- radiation-induced thermoelectric sensitivity
- radiation-induced absorption
- radio-luminescence
- thermal conductivity decrease
- volume changes (swelling)
- .....

→ Feasibility of in vessel magnetic diagnostics to be further analysed

Current concept for implementation of diagnostic front-end:

durable components; retracted mounting positions

- **In front of blanket and divertor:**
  - No diagnostic components
- **Within blanket area:**
  - Microwave antennae+waveguides
  - Cassettes and tube-like penetrations for interferometry/polarimetry, spectroscopy, neutron+gamma
- **Behind blanket:**
  - Optical mirrors and beam paths
  - magnetic sensors with cabling t.b.d. (depending on lifetime perspective)
- **Port plugs:**
  - penetrations
  - mirror labyrinths
- **Within divertor target:**
  - No diagnostic components
  - tube-like access? (low power region, t.b.d.)
- **Behind divertor target:**
  - implementation of thermo-current measurement t.b.d.
  - need for lines of sight t.b.d.



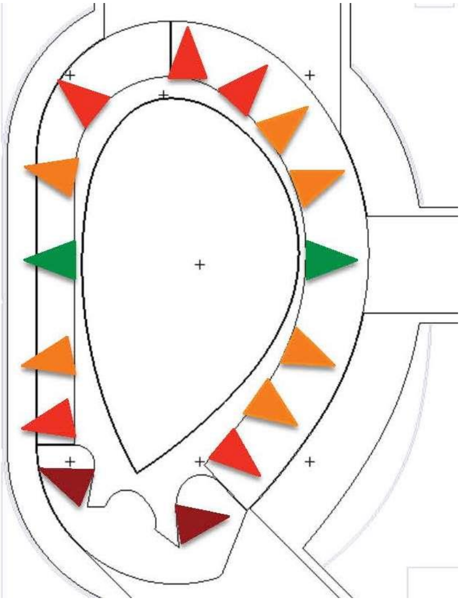
Main diagnostic methods on DEMO (1):
microwave diagnostics

Diagn. methods:	➢ reflectometry ➢ ECE
Control tasks to be covered:	➢ n_e profile in gradient region ➢ T_e profile ➢ plasma position and shape ➢ plasma instabilities
Implementation:	➢ > 100 antennea distributed toroidally and poloidally (LFS, HFS, upper, x-point) ➢ integration of antennae and waveguides in slim modules between blanket boxes ➢ routing of waveguides to upper port (attached to „blanket bananas“) ➢ Microwave feedthroughs or ceramic vacuum breaks near upper port

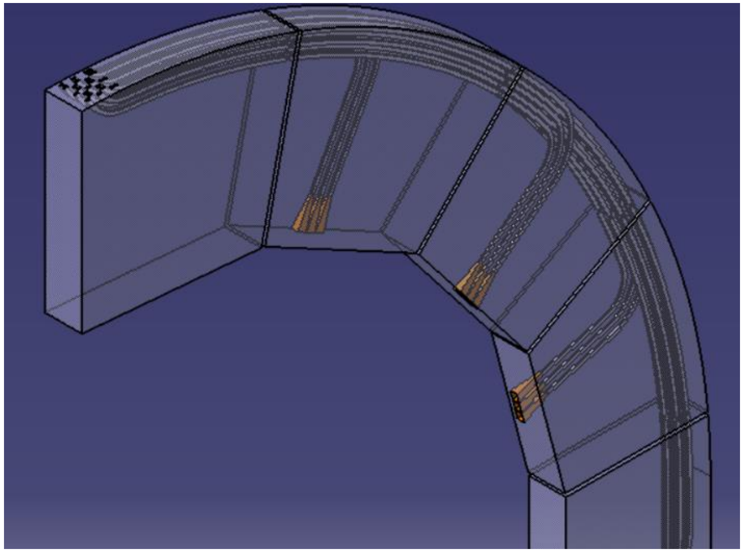
In case of 2 slim sectors (< 25 cm each), the effect on TBR would be ΔTBR < 0.01

A. Silva,
A. Malaquias,
C. Sozzi,
G. de Masi, et al.

Feasibility of reflectometry on top and bottom side is not clear



Integration study of MW antennae

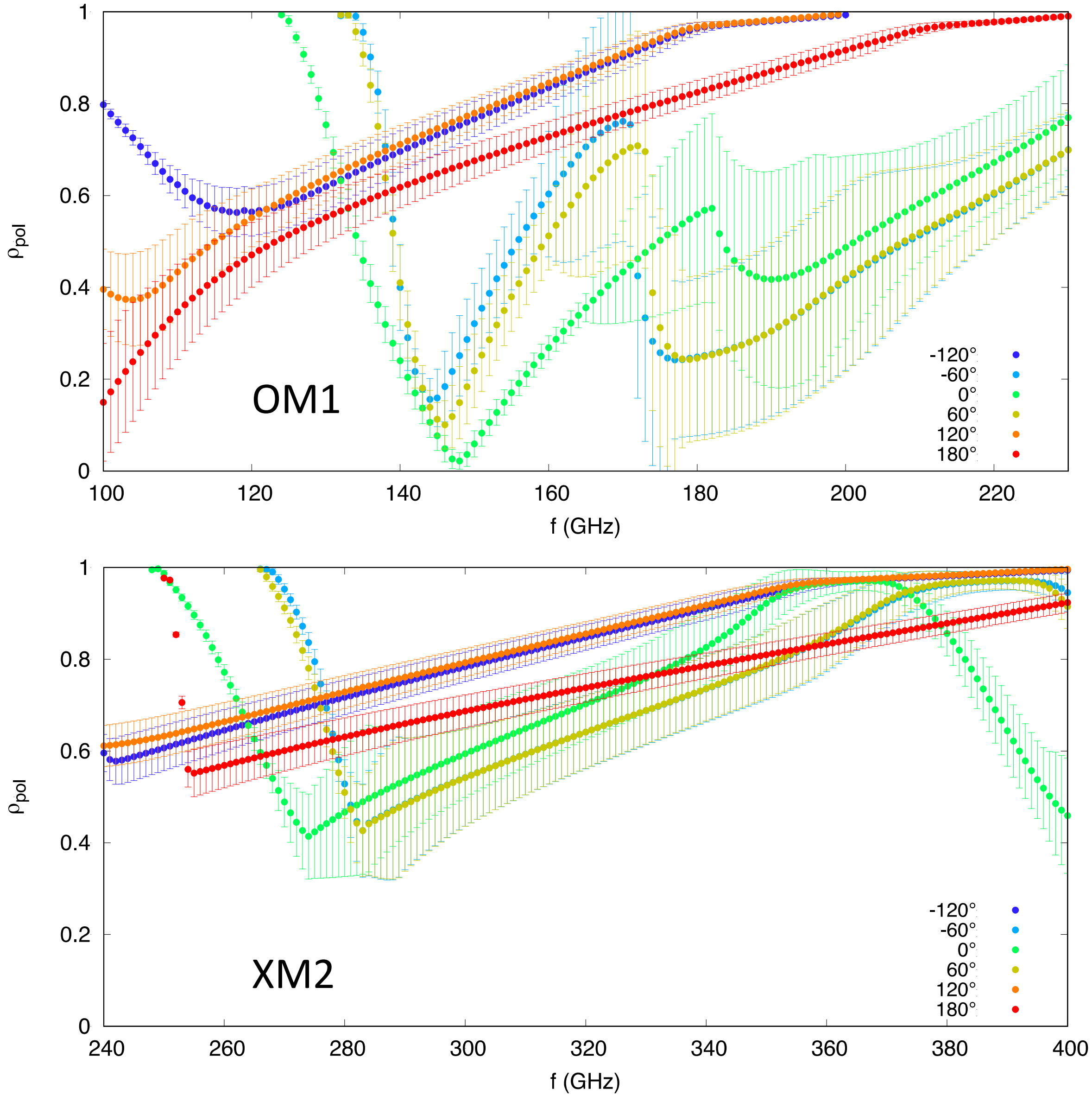


Example: spatial resolution of ECE measurements
(Measurement of electron temperature, and instability detection)

Simulation of microwave propagation under DEMO conditions

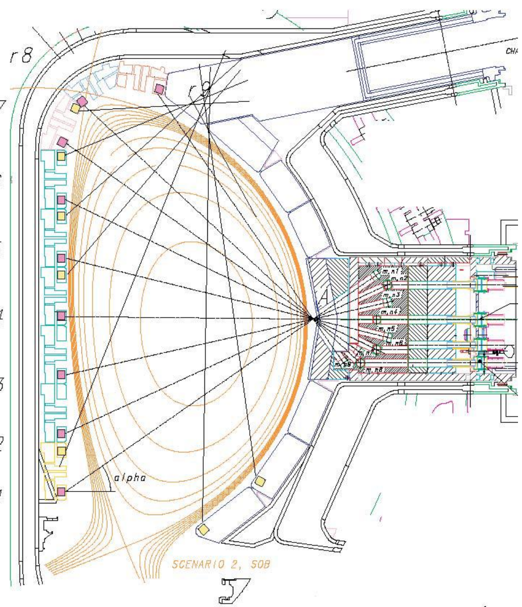
(C. Sozzi et al., 2015)

- The radial width $\Delta\rho_{pol,1/e}$ of EC emission is shown as a function of the normalised radius ρ_{pol} and the poloidal angle
- **Good spatial resolution at LFS for 0° observation**
- A moderate spatial resolution ($\Delta\rho < 0.2$) can be achieved all over the plasma by
  - using **antennae mounted in different poloidal angles**
  - using both **ordinary mode at base frequency (OM1)** and **2nd harmonic extraordinary mode (XM2)**

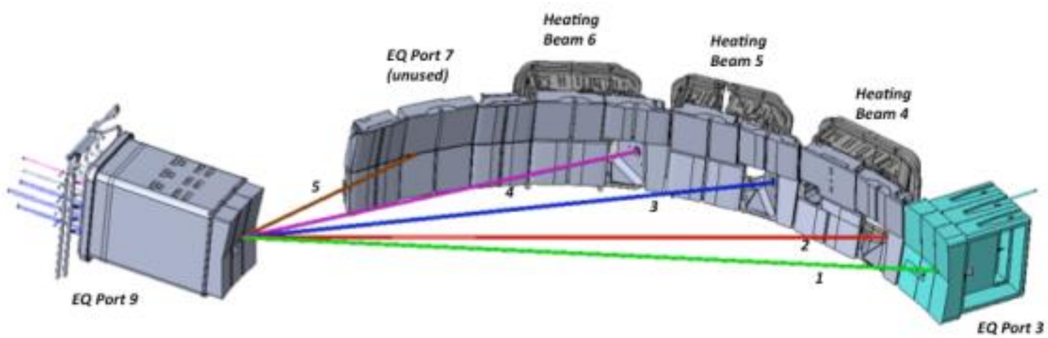


Main diagnostic methods on DEMO (2):
IR polarimetry/interferometry

Diagn. methods:	➢ IR polarimetry, combined with interferometry
Control tasks to be covered:	➢ n_e profile in core plasma (r/a < 0.7) ➢ plasma position and shape (contribute) ➢ plasma instabilities (contribute)
Implementation:	➢ modular system, ~ 20 beams in core region (radial profile, redundancy) ➢ all beams entering at eq. ports; IR windows at port plate or bioshield t.b.d. ➢ lasers and detectors located behind bio-shield (facilitate maintenance) ➢ first mirror and retro-reflectors behind blanket (inboard or outboard, tbd), mounted to blanket or Vacuum Vessel (VV) tbd

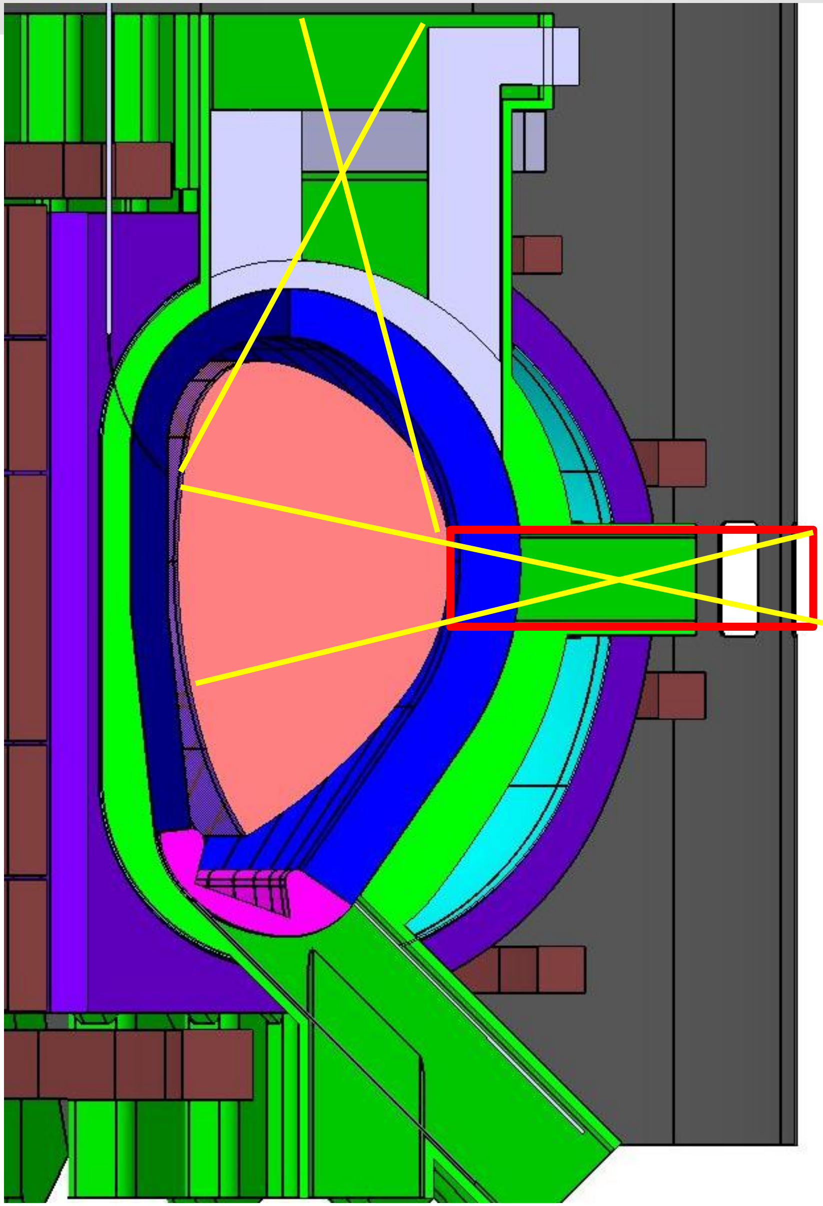


→ work started in 2016, ITER-like system envisaged.



Main diagnostic methods on DEMO (3):
neutron/gamma diagnostics

- Diagnostics methods:
  - Neutron and gamma flux measurements
  - Neutron and gamma spectroscopy
- Control tasks to be covered:
  - fusion power density profile + total power
  - plasma position and shape (contribute)
  - fuel ion temperature (from neutron and gamma spectra)
  - fuel ion density (from neutron spectra)
  - fault detection, i.e., anomalies in planned plasma scenario (fuel ions reactions with impurities/wall) from gamma spectra
- Integration issues:
  - Equatorial Port system: ~ 10-20 beams in core region (radial profile for r/a < 0.4)
  - Vertical Port system: ~ 20 – 30 beams covering the major part of the plasma cross section
  - detectors located behind bio-shield (facilitate maintenance)



Approaches for DEMO power exhaust control

- Two contradicting requirements for power exhaust control are driving the requirements:
  - q_{div} < divertor wall limit – margin
  - P_{sep} > P_{LH} + margin
- Actuators for divertor control: impurity injection for plasma core radiation and for divertor radiation

Diagnostic approaches under consideration for power exhaust control:

1) Measurement of q_{div} and P_{sep}

   indirect method; small differences of big numbers to be measured
   - $P_{sep} = P_{heat} - P_{rad,core}$
   - $q_{div} \sim (P_{sep} - P_{rad,div}) / A_{HHF}$
2) Thermography on divertor target

   strong background radiation in detached plasma
   - needs lines of sight with angle > 30 degrees against divertor target
3) Measurement of divertor thermo-current

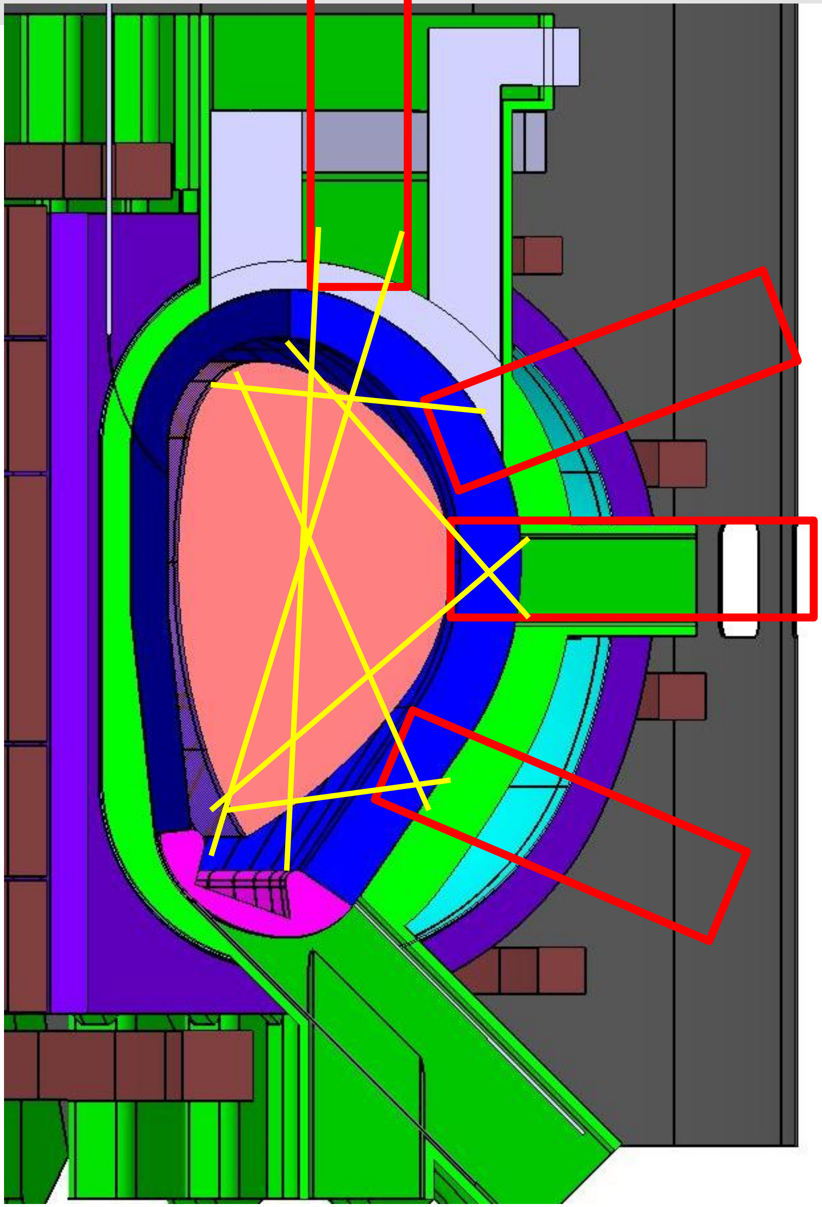
   isolated mounting of divertor, or low voltage only
   - based on sheath voltage due to plasma-wall contact
4) Measurement of spatial distributions in divertor plasma (spectroscopy / P_{rad})

   spatial resolution?
   - T, v, location of ionisation front, fluxes(?)
   - check for X-point MARFE (upper limit of detachment)
   - assuming that full detachment is sufficient for divertor protection
5) Spectroscopic measurement of W erosion flux at divertor strike-point

   spatial resolution?
   - control (limitation) of W erosion of the divertor target needed
   - erosion is related to flux and energy distribution, like q_{div}
   - assuming that vanishing W flux is sufficient for divertor protection (non-monotonic relation?)

Main diagnostic methods on DEMO (4):
spectroscopic and radiation measurements

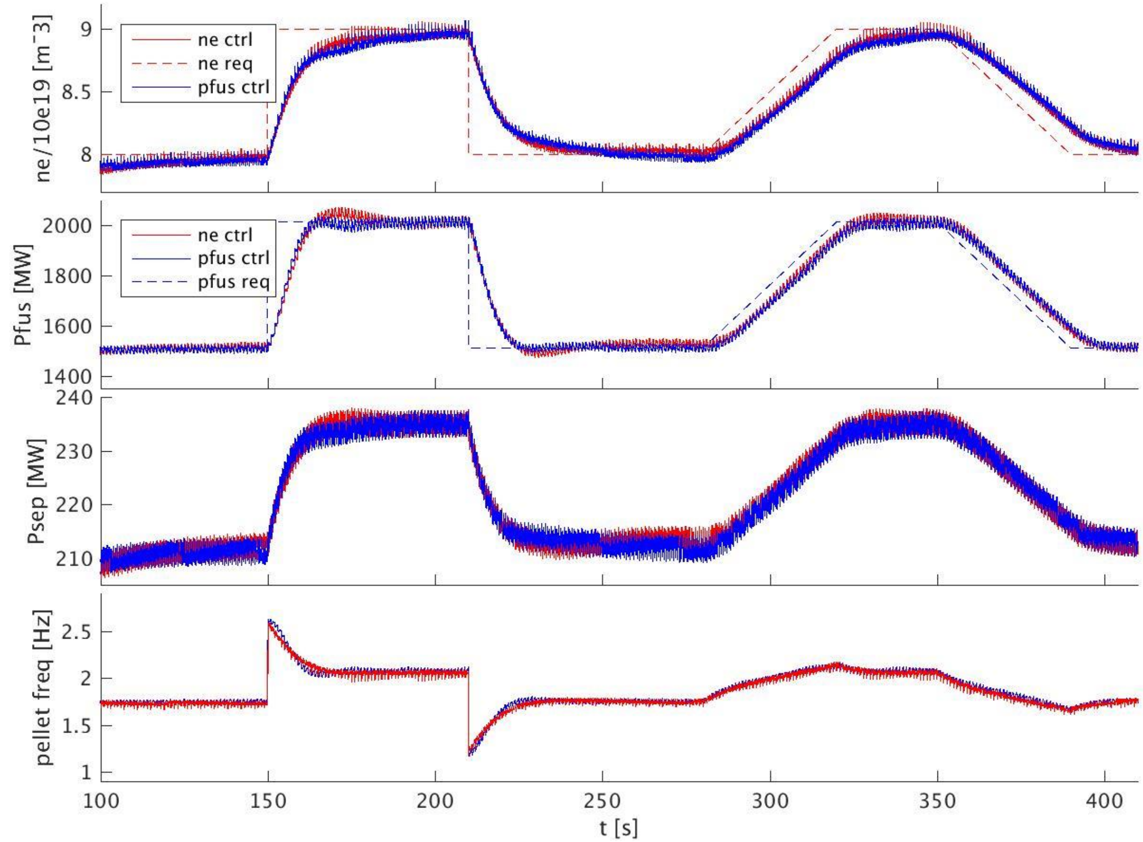
- Diagnostics methods:
  - spectroscopy from X-ray to IR
  - radiation (power) measurement
  - divertor thermography
- Control tasks to be covered:
  - impurity control
  - radiation power control
  - wall surface temperature control
  - divertor detachment control
  - other quantities t.b.d.
- Integration issues:
  - > 150 measurements, mostly by individual narrow lines of sight
  - first mirrors behind blanket (low neutron fluence)
  - long ducts with opening L/D > 30 (low erosion/deposition)
  - secondary mirrors in vessel to guide light to ports
  - windows or vacuum penetrations at ports
  - detectors located behind bio-shield (facilitate maintenance)



Kinetic control simulations (plasma core)

- Promising new development: ASTRA coupled with Simulink
- Actuators such as pellet injection, gas injection, pumping and heating implemented
- Diagnostics effects currently simulated via noise (to be improved)
- First quantitative results: density follows a requested change with ~ 10 sec delay
- Future work:
  - progress to implement more and more realistic features (e.g. diagnostic + actuator accuracy)
  - extend the model towards power exhaust and MHD

H. Zohm et al.



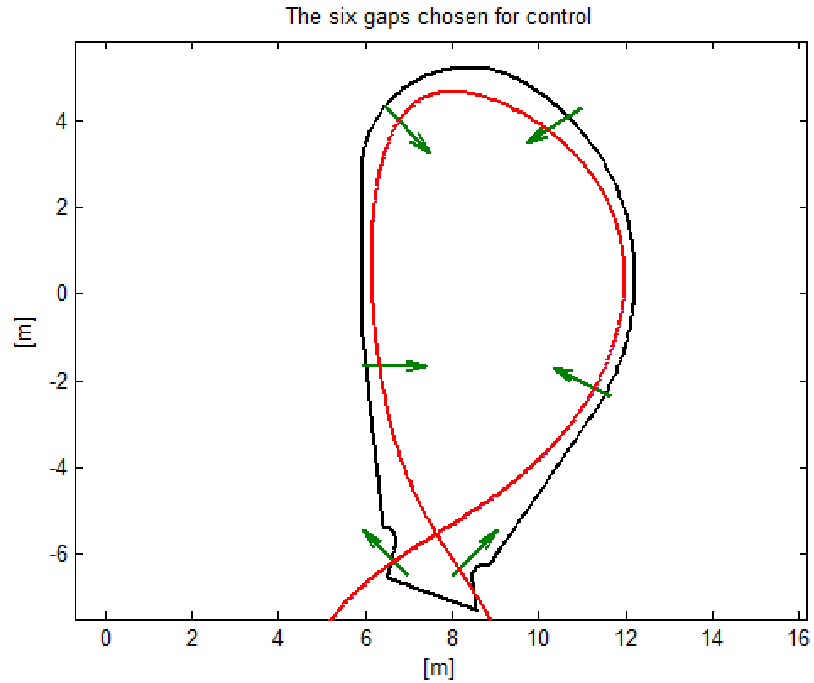
Control variable:

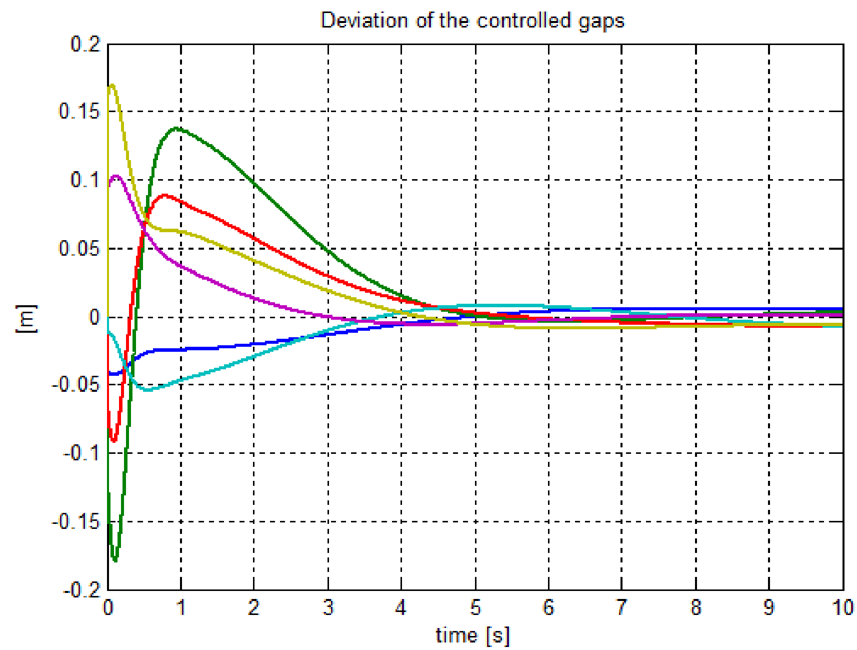
- n_e
- P_{fus}

Plasma shape (gap) control

(M. Ariola et al.)

- Control simulations for plasma shape control based on magnetic sensors have been conducted
- Approximate treatment of the blanket (eddy current shielding) included
- Modelled plasma disturbance:
  - Big „ELM“ (step function)
- First quantitative results:
  - overshooting 5-15 cm
  - settling time ~ 2-5 sec
- Future work:
  - step-by-step model refinement to include realistic DEMO properties:
    - diagnostics other than magnetics
    - PF coil properties (limited voltage and power)
    - eddy current shielding by blanket and VV





DEMO control quantities and operational limits

Control quantity	**Operational limits**
Plasma current	q_{95} limit (safety factor)
Plasma density (edge rather than core)	density limit
Plasma radiation, impurity mixture, Z_{eff}	radiation limit LH threshold
Fusion power	local wall loads (FW and div.) LH threshold
Divertor detachment and heat flux control	LH threshold local wall loads in divertor
Plasma position and shape, incl. vertical and horizontal stability	local wall loads (FW and div.) max. Δz / VDE disruption
(MHD) plasma instabilities	various ($\rightarrow$ disruptions)
Plasma pressure	beta limit

Outlook: ongoing and future R&D

- systems engineering and integration
  - development of requirements
  - assessment of diagnostic and actuator properties
  - diagnostic integration studies
  - refinement of the plasma scenario (controllability)
- control simulations:
  - burn control
  - exhaust control
  - position and shape control
  - MHD control
- diagnostic R&D
  - performance of the selected diagnostics under DEMO conditions
  - component lifetime assessment
  - study of alternative diagnostic methods

Conclusions

- Diagnostic implementation on DEMO is limited by
  - degradation of front-end components by ionising radiation, erosion, deposition
    - retracted mounting of front-end components
    - restrict to selection of robust methods only
  - need to maximise the Tritium breeding rate (low space available for diagnostics)
  - remote maintenance and integration issues
  - integration and performance of initial diagnostic suite for DEMO is under study
- Actuator properties on DEMO are limited as well
  - limited amount of auxiliary power
  - ex-vessel coil system only, eddy current shielding by blanket and vacuum vessel
- Consequently the DEMO control system will only provide limited performance
  - Feasibility of reliable control may limit the operational space for the plasma scenario
- Novel integrated control techniques may (partially) compensate the shortfalls on the diagnostic/actuator side
- Realistic (conservative) approach and sufficient control margins are needed in machine design and plasma scenario in order to achieve high reliability of DEMO operation

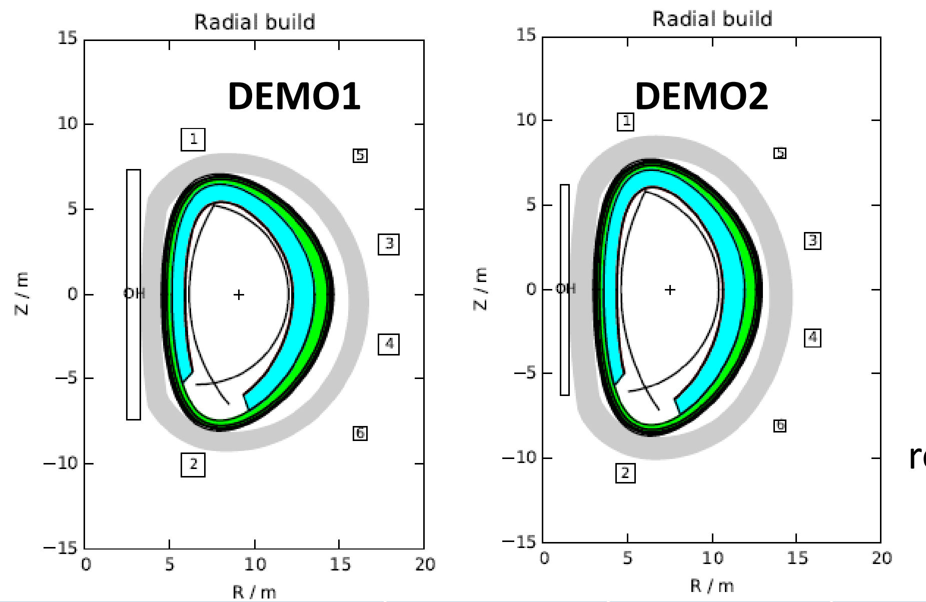
Spare slides

Preliminary DEMO Design Choices under evaluation

(after G. Federici)

Design features (near-term DEMO):

- 2000 MWth~500 Mwe
- Pulses > 2 hrs
- Single Null water cooled divertor
- PFC armour: W
- LTSC magnets Nb_3Sn (grading)
- B_{max} conductor ~12 T
- RAFM (EUROFER) as blanket structure material
- Vacuum vessel made of AISI 316
- Blanket vertical RH / divertor cassettes
- Lifetime: starter blanket: 20 dpa (200 appm He); 2nd blanket 50 dpa; divertor: 5 dpa (Cu)



Under revision

Open Choices:

- Operating plasma scenario
- Breeding blanket design concept selection
- Primary Blanket Coolant/ BoP
- Protection strategy first wall (e.g., limiters)
- Divertor configurations (SN, DN, advanced)
- Number of TF coils

	ITER	DEMO1 (2015) A=3.1	DEMO2 (2015) A=2.6
R_0 / a (m)	6.2 / 2.0	9.1 / 2.9	7.5 / 2.9
K_{95} / δ_{95}	1.7 / 0.33	1.6 / 0.33	1.8 / 0.33
A (m^2)/ Vol (m^3)	683 / 831	1428 / 2502	1253 / 2217
H non-rad-corr / β_N (%)	1.0 / 2.0	1.0 / 2.6	1.2 / 3.8
P_{sep} (MW)	104	154	150
P_F (MW) / P_{NET} (MW)	500 / 0	2037 / 500	3255 / 953
I_p (MA) / f_{bs}	15 / 0.24	20 / 0.35	22 / 0.61
B at R_0 (T)	5.3	5.7	5.6
$B_{max, conductor}$ (T)	11.8	12.3	15.6
BB i/b / o/b (m)	0.45 / 0.45	1.1 / 2.1	1.0 / 1.9
Av NWL MW/m^2	0.5	1.1	1.9

First wall heat loads due to disruptions on DEMO

Heat impact factor:

$$\eta = W_{th}/A_{eff}/\sqrt{t}$$

Critical heat impact factor (melting):

$$\eta_{crit} = (T_{melt} - T_{op})\sqrt{\pi\lambda\rho c/4}$$

using

T_{melt}	=	melt temperature	
T_{op}	=	operational temperature	
λ	=	heat conductivity (W/mK)	
ρ	=	mass density	(kg/m^3)
c	=	heat capacity	(J/kgK)

Data for tungsten:

λ	=	170	W/mK
ρ	=	19300	kg/m^3
c	=	138	J/kgK

Estimating DEMO disruption loads
(based on the ITER approach / thermal quench):

In a mitigated disruption, 50% of the thermal energy is radiated within the quench time t_{TQ}

$t_{TQ,DEMO}$	=	1 ms (time constant ~ a)
W_{therm}	~	1000 MJ
peaking	~	factor 4

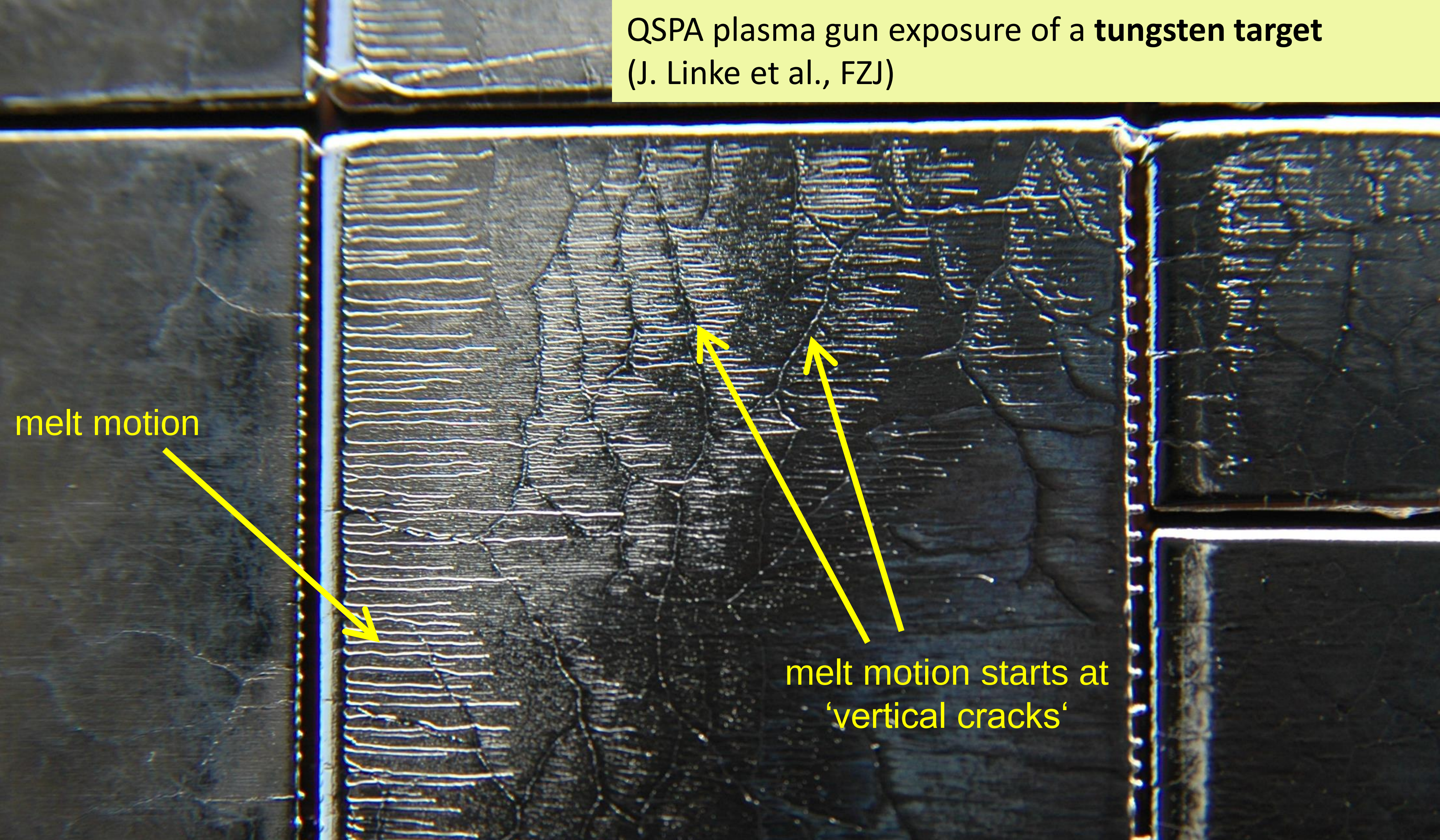
yields:

$\eta_{TQ,DEMO}$	**~**	**30 $MJ/m^2/s^{0.5}$**
η_{crit}	~	50 $MJ/m^2/s^{0.5}$ (melting)
η_{crit}	~	3-6 $MJ/m^2/s^{0.5}$ (cracking)

Even mitigated disruptions could cause large area crack damage on DEMO components

→ disruption avoidance should have very high priority on DEMO

→ data on the number of permissible heat transients depending on pulse energy are needed for DEMO FW and divertor components (input for DEMO design)



QSPA plasma gun exposure of a **tungsten target**
(J. Linke et al., FZJ)

High energy disruptions on DEMO

- High risk of major wall damage, in-vessel inspection may be required before restart
- Reactor operation with high availability requires very low disruption rate (< 1 / fpy)

→ large improvement of control reliability needed as compared to JET average

Radiation effects on diagnostic components

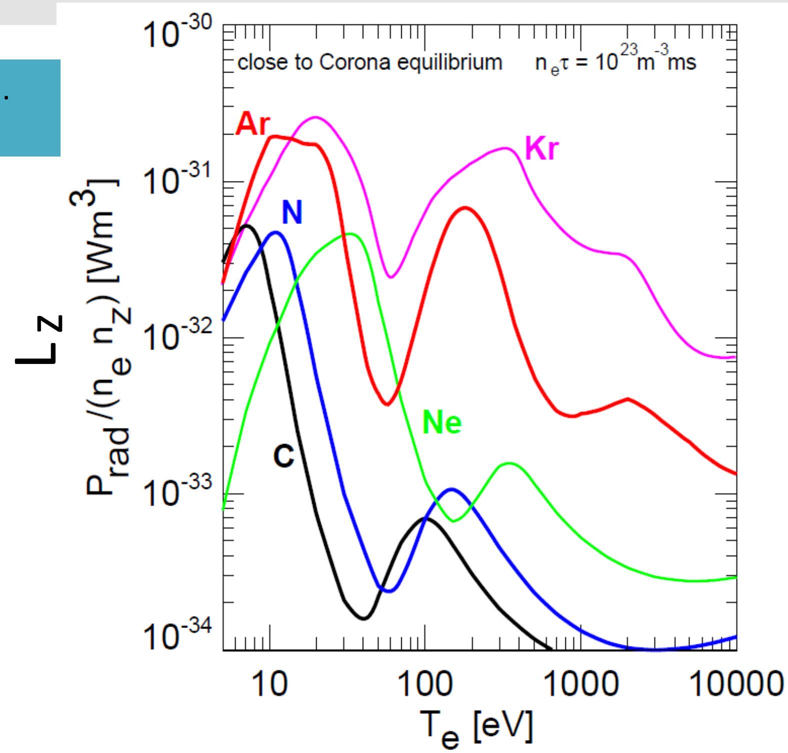
Neutron fluence and activation on DEMO behind the blanket will be comparable to the situation near the ITER first wall

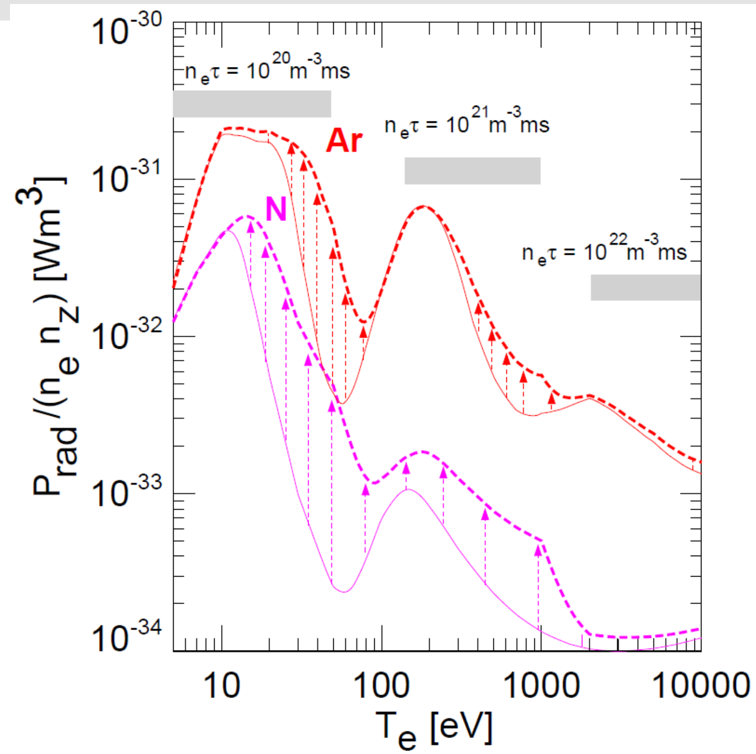
(table from:
G. Vayakis, IO,
ITER_D_2UYLBG)

Effect	Symbol	Explanation
Radiation-induced conductivity	RIC	Electrical conductivity increases due to the excitation of electrons into the conduction band.
Radiation-induced electrical degradation	RIED	Electrical conductivity increases due to radiation and electric field enhanced defect aggregation.
Thermal conductivity decrease	–	Thermal conductivity decreases leading to temperature increases.
Volume changes	–	Materials swell, or in some cases shrink
Radiation-Induced Electromotive Force	RIEMF	Nuclear reactions in the sensor materials induce net current in the sensor circuit
Thermoelectric Electromotive Force	TIEMF	Parasitic thermocouple action driven by nuclear-heating
Radiation-induced thermoelectric sensitivity	RITES	Additional parasitic thermocouples generated by non-uniform material damage and transmutation
Radiation-enhanced diffusion	–	Enhanced diffusion occurs in insulating materials due to the possible existence of different charge states for defects and impurities.
Radiation-induced absorption	RIA	Optical absorption increases due to the production of defect related absorption bands, leading to light transmission loss.
Radioluminescence or radiation-induced emission	RL or RIE	Light emission due to excitation of defects and impurities.

Exhaust control and radiation instability

A. Kallenbach et al.
PPCF 2013

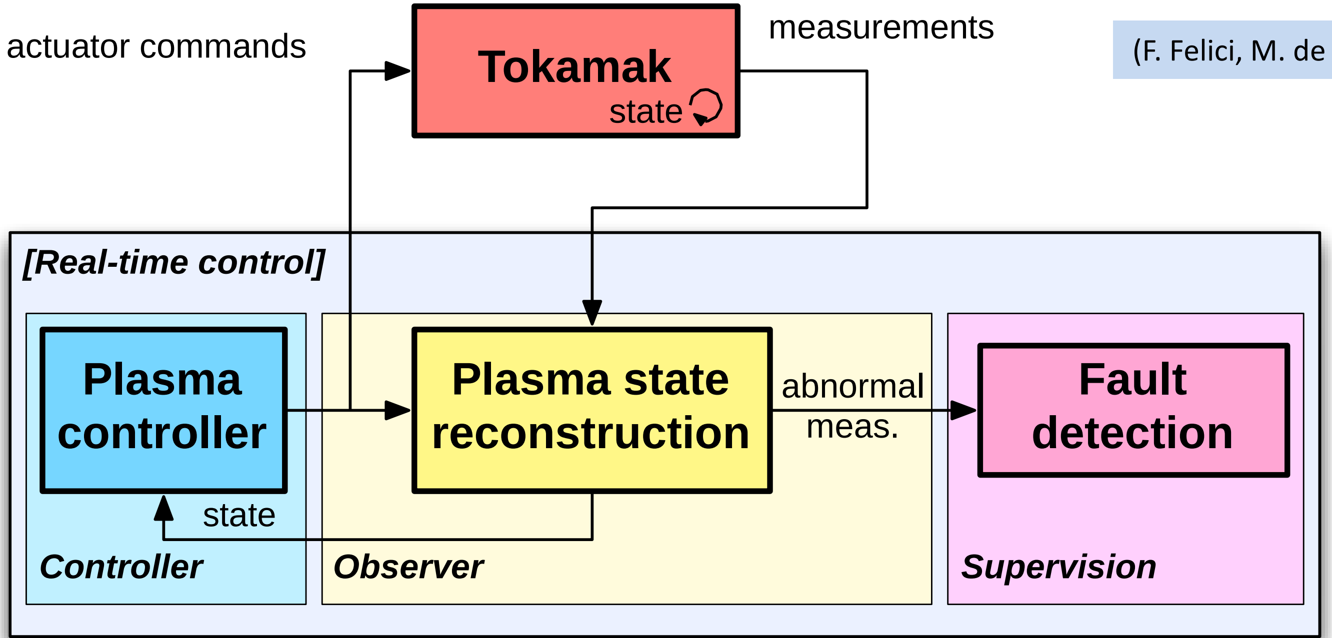




- Radiation level to be adjusted using seed impurities
- Radiation instability: L_Z rises when T_e decreases, and alpha power decreases
  - Measurement of plasma radiation only available at low performance
  - Only low auxiliary heating power available to counteract a radiation collapse
  - Plasma density can only be slowly reduced (limited by pumping speed)
- Thorough stability analysis / dynamic simulation needed before concluding on the scenario

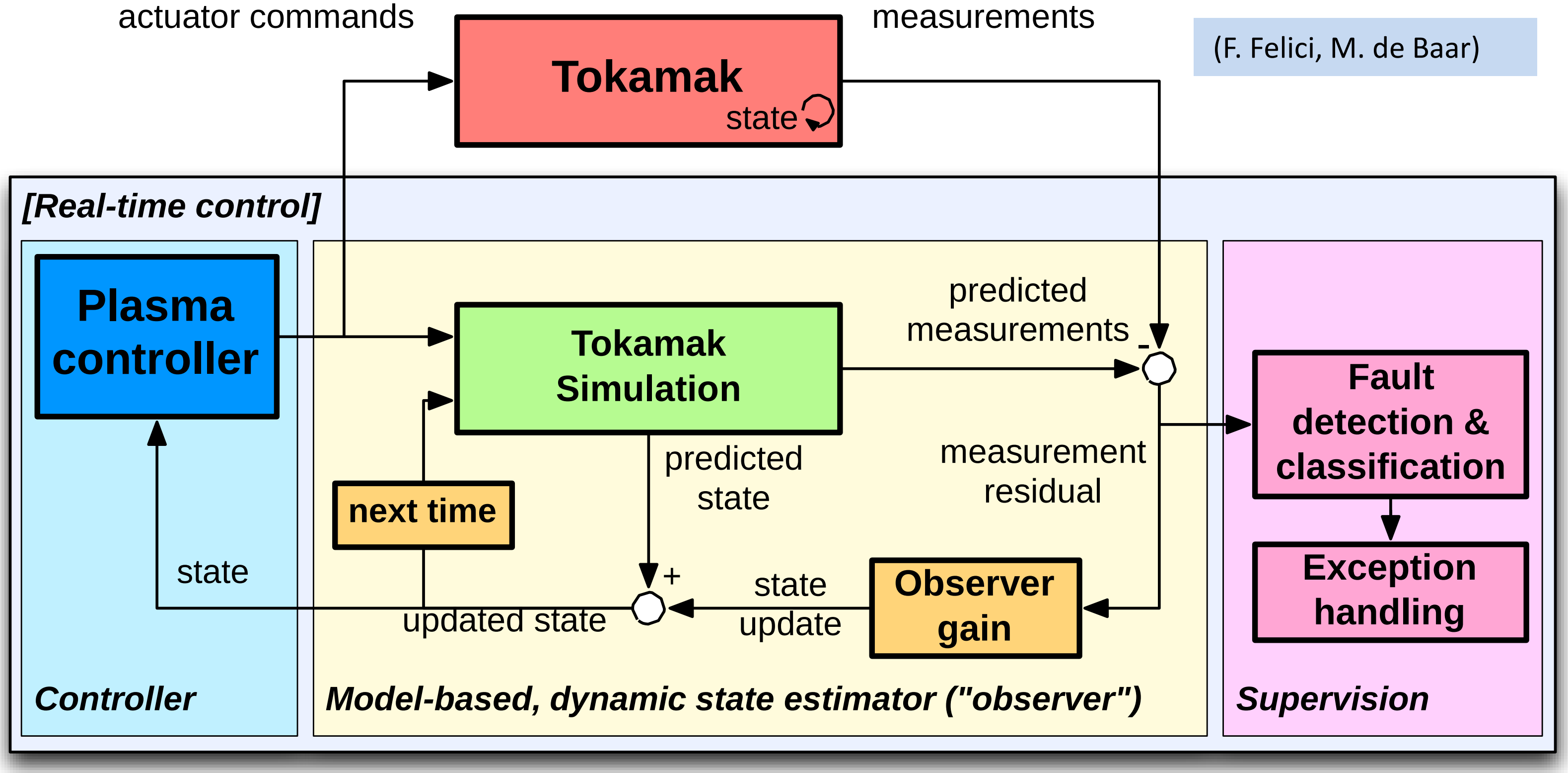
Integrated data analysis and control of a tokamak plasma

(F. Felici, M. de Baar)



- Plasma controller: perform control actions based on full plasma state knowledge
- Plasma state reconstruction: derive plasma state by merging measurements from several diagnostics
- Fault detection: classify unexpected measurements (e.g. off-normal events, faulty signals)
- Diagnostic redundancy in number of channels and number of methods facilitates handling of faults (the better the model, the less measurements are needed)

Advanced control of a tokamak plasma



- Run tokamak simulation in parallel with plasma evolution
- Correct simulated state estimate based on difference between predicted and true measurements
- Detection & classification of excessive discrepancies
- The plasma controller may initiate fast rampdown or disruption mitigation if a discrepancy cannot be resolved otherwise